American Nuclear Society

methods for determining neutron fluence in BWR and PWR pressure vessel and reactor internals

an American National Standard

REAFFIRMED

October 11, 2016

ANSI/ANS-19.10-2009; R2016

This standard has been reviewed and reaffirmed with the recognition that it may reference other standards and documents that may have been superseded or withdrawn. The requirements of this document will be met by using the version of the standards and documents referenced herein. It is the responsibility of the user to review each of the references and to determine whether the use of the original references or more recent versions is appropriate for the facility. Variations from the standards and documents referenced in this standard should be evaluated and documented.
This standard does not necessarily reflect recent industry initiatives for risk informed decision-making or a graded approach to quality assurance. Users should consider the use of these industry initiatives in the application of this standard.

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American National Standard Methods for Determining Neutron Fluence in BWR and PWR Pressure Vessel and Reactor Internals

Secretariat
American Nuclear Society

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Working Group ANS-19.10

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American National Standard

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Foreword

(This Foreword is not part of American National Standard "Methods for Determining Neutron Fluence in BWR and PWR Pressure Vessel and Reactor Internals," ANSI/ANS-19.10-2009.)

It is the intent of this American National Standard to provide guidance for the evaluation of pressurized water reactor (PWR) and boiling water reactor (BWR) pressure vessel and reactor internals fast ($E > 1.0$ MeV) neutron fluence. This standard outlines the attributes of the method(s), the necessary types of data, the required benchmarking of the method, and the necessary steps in performing the calculations. The method(s) described herein require both experimentally measured vessel dosimetry data and corresponding fast neutron fluence calculations to perform the benchmark. This also allows the user to determine the existence of a bias in the calculated values and to quantify its magnitude. Likewise, the information needed for the benchmark allows the quantification of uncertainties. The method or methods described in this standard calculates a best-estimate value that is suitable (and acceptable) for use in applications related to *Code of Federal Regulations*, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements." The intended applications are for American-made PWRs and BWRs.

Compliance with the intent of this standard can be demonstrated by meeting the following two requirements:

(1) The calculation must be validated as described in Sec. 5 of this standard;

(2) the validation must be based on a qualified data set from dosimetry measurements performed as described in Sec. 4 of this standard.

This standard might reference documents and other standards that have been superseded or withdrawn at the time the standard is applied. A statement has been included in the reference section that provides guidance on the use of references.

This standard was developed by Working Group ANS-19.10 of the American Nuclear Society. At the time of the standard's completion, the following members participated in the current version:

L. Lois (Chair), *U.S. Nuclear Regulatory Commission*
J. F. Carew (Secretary), *Brookhaven National Laboratory*

J. M. Adams, *National Institute of Standards and Technology*
S. Anderson, *Westinghouse Electric Company, LLC*
R. J. Cacciapouti, *Individual*
A. Haghighat, *University of Florida*
W. C. Hopkins, *Individual*
J. R. Worsham, *AREVA*
M. Mahgerefteh, *Exelon Corporation*
S. P. Baker, *Transware Enterprises*
J. C. Wagner, *Oak Ridge National Laboratory*
Y. Orechwa, *U.S. Nuclear Regulatory Commission*
R. C. Little, *Los Alamos National Laboratory*

The membership of Subcommittee ANS-19, Reactor Physics Standards, at the time of its review and approval of this standard was as follows:

D. M. Cokinos (Chair), *Brookhaven National Laboratory*
C. T. Rombough (Secretary), *CTR Technical Services, Inc.*

W. H. Bell, *American Institute of Chemical Engineers*
M. Brady-Raap, *Pacific Northwest National Laboratory*
D. J. Diamond, *Brookhaven National Laboratory*
J. Katakura, *Japan Atomic Energy Research Institute*
E. R. Knuckles, *Florida Power and Light*
R. C. Little, *Los Alamos National Laboratory*
L. Lois, *U.S. Nuclear Regulatory Commission*
R. D. Mosteller, *Los Alamos National Laboratory*
B. Rouben, *12 & 1 Consulting*
A. Weitzberg, *Individual*

Consensus Committee N17, Research Reactors, Reactor Physics, Radiation Shielding and Computational Methods, had the following membership at the time it reviewed and approved the standard:

T. M. Raby (Chair), *National Institute of Standards and Technology*
A. Weitzberg (Vice Chair), *Individual*

S. Anderson, *Westinghouse Electric Company, LLC*
W. H. Bell, *American Institute of Chemical Engineers*
(Alt. R. D. Zimmerman, *American Institute of Chemical Engineers*)
R. E. Carter, *Individual*
D. M. Cokinos, *Brookhaven National Laboratory*
M. L. Corradini, *National Council on Radiation Protection & Measurement*
B. Dodd, *Health Physics Society*
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N. E. Hertel, *Georgia Institute of Technology*
C. J. Heysel, *McMasters University*
W. A. Holt, *Individual*
W. C. Hopkins, *Individual*
M. A. Hutmaker, *U.S. Department of Energy*
A. C. Kadak, *Massachusetts Institute of Technology*
L. I. Kopp, *Individual*
P. M. Madden, *U.S. Nuclear Regulatory Commission*
(Alt. A. Adams, *U.S. Nuclear Regulatory Commission*)
J. F. Miller, *Institute of Electrical and Electronics Engineers*
J. E. Olhoeft, *Individual*
R. E. Pevey, *University of Tennessee, Knoxville*
W. J. Richards, *National Institute of Standards and Technology*
C. T. Rombough, *CTR Technical Services, Inc.*
T. R. Schmidt, *Sandia National Laboratories*
S. H. Shepherd, *Individual*
A. O. Smetana, *Savannah River National Laboratory*
R. R. Tsukimura, *Aerotest Operations*
A. R. Veca, *General Atomics*
S. H. Weiss, *National Institute of Standards and Technology*
(Alt. T. J. Myers, *National Institute of Standards and Technology*)

Contents

Methods for Determining Neutron Fluence in BWR and PWR Pressure Vessel and Reactor Internals

1 Introduction and scope

1.1 Introduction

This standard is intended for use by

(1) those involved in the determination of the pressure vessel fluence to satisfy the requirements of *Code of Federal Regulations*, Title 10, Part 50, Section 61 (10 CFR 50.61) [1][1)]; 10 CFR 50.61, Appendix G [2]; and 10 CFR 50.61, Appendix H [3];

(2) those involved in the determination of material properties of irradiated reactor vessel and reactor internals;

(3) regulatory agencies in the evaluation of licensing actions concerning the material properties of irradiated pressure vessels and irradiated reactor internals.

1.2 Scope, purpose, and application

This standard provides a procedure for the evaluation of the best-estimate fast ($E > 1.0$ MeV) neutron fluence in the annular region between the core and the inside surface of the vessel, through the pressure vessel and the reactor cavity, between the top and bottom of the active fuel given the neutron source in the core. This evaluation employs both fast neutron flux computations and measurement data from in-vessel and cavity dosimetry, as appropriate. The standard applies to both U.S.-designed pressurized water reactors (PWRs) and boiling water reactors (BWRs).

2 Definitions

The following definitions apply for the purposes of this standard. Other specialized terms conform to *Glossary of Terms in Nuclear Science and Technology* [4].

benchmark experiment: A well-defined set of physical experiments with results judged to be sufficiently accurate for use as a calculational reference point. The judgment is made by a group of experts in the subject area.

best-estimate fluence: The most accurate value of the fluence based on all available measurements, calculated results, and adjustments based on bias estimates, least-squares analyses, and engineering judgment.

calculational methodology: The mathematical equations, approximations, assumptions, associated parameters, and calculational procedure that yield the calculated results. When more than one step is involved in the calculation, the entire sequence of steps comprises the "calculational methodology."

code benchmark: Comparison to the results of another code system that has been previously validated against the benchmark experiment(s).

continuous-energy cross-section data: Cross-section data that are specified in a dense pointwise manner that comprises the energy range.

dosimeter reaction: A neutron-induced nuclear reaction with a product nuclide having sufficient activity to be measured and related to the incident neutron fluence.

least-squares adjustment procedure: A method for combining the results of neutron transport calculations and the results of dosimetry measurements that provides an optimal estimate of the fluence by minimizing, in the least-squares sense, the calculation-to-measurement differences.

multigroup cross-section data: Cross-section data that have been determined by averaging the continuous-energy cross-section data over discrete energy intervals using specified weighting functions.

[1)]Numbers in brackets refer to corresponding numbers in Sec. 7, "References."